NON-PUBLIC?: N
ACCESSION #: 9303040255
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 2 PAGE: 1 OF 4

DOCKET NUMBER: 05000311

TITLE: Manual Rx Trip From 100% Power Upon Trip of Both Steam
Generator Feed Pumps
EVENT DATE: 01/28/93 LER #: 93-002-00 REPORT DATE: 02/26/93

OTHER FACILITIES INVOLVED: DOCKET NO: 050001

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. J. Pollack - LER Coordinator TELEPHONE: (609) 339-2022

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: CON MANUFACTURER: H015
B EC AHU A500

REPORTABLE NPRDS: Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 1/28/93, the low suction pressure alarm for both Steam Generator Feedwater Pumps (SGFPS) was received. Both pumps tripped on low suction pressure. A manual reactor trip was then initiated. The Unit was stabilized in Hot Standby. Prior to the loss of the SGFPS, a technician was connecting a brush recorder to the inputs and output of the Steam Generator Feed Pump (SGFP) Master Controller to troubleshoot observed spikes on all 4 Steam Generator feed flow instrument channels. The root cause of this event is equipment failure. The SGFP master controller module test jack was sufficiently loose to allow the weight of the troubleshooting lead to bring the threads of the troubleshooting test lead into contact with the module chassis. This created an electrical short between the controller signal common and chassis ground, which

produced an erroneous maximum speed demand signal. Flow through the pumps then increased resulting in the low suction pressure and the pump trips. The loose SGFP master controller test jack was repaired and other jacks in the SGFP speed control loop circuitry were inspected and repaired as required. The other Hagan protection channel test connections (both Salem Units) are being checked during the protection channel monthly functional surveillance. The Hagan process channel test connections, both Salem Units, will be checked for looseness. Engineering is reviewing whether changes to the test jacks are needed to prevent event recurrence.

END OF ABSTRACT

TEXT PAGE 2 OF 4

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

IDENTIFICATION OF OCCURRENCE:

Manual Reactor Trip From 100% Power Upon Trip of Both Steam Generator Feedwater Pumps

Event Date: 1/28/93

Report Date: 2/26/93

This report was initiated by Incident Report No. 93-128.

CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 100% - Unit Load 1170 MWe

At approximately 1341 hours on January 28, 1993, an Instrumentation and Controls (I&C) technician was connecting a brush recorder to the inputs and output of the Steam Generator Feed Pump (SGFP) Master Controller, 2FC500H, to troubleshoot observed spikes on all four (4) Steam Generator feed flow instrument channels. To minimize the possibility of losing SGFP speed control, the controller was placed in "manual". The brush recorder was connected in accordance with procedure SC.IC-GP.ZZ-0006(Q), Controls Equipment - Troubleshooting.

DESCRIPTION OF OCCURRENCE:

At approximately 1347 hours, on January 28, 1993, the low suction pressure alarm for 22 SGFP was received and the pump tripped on low suction pressure. Actions were initiated for loss of a SGFP. It was then observed that 21 SGFP had also tripped on low suction pressure and at 1347 hours (same day). With the loss of both SGFPS, a manual Reactor trip was initiated. The Unit was then stabilized in MODE 3 (Hot Standby). At 1518 hours (same day) the Nuclear Regulatory Commission was notified of the manual actuation of the Reactor Protection System {JC}, in accordance with the requirements of 10CFR 50.72 (b) (2) (ii).

APPARENT CAUSE OF OCCURRENCE:

The root cause of this event is equipment failure. The SGFP master controller module test jack was sufficiently loose to allow the weight of the troubleshooting lead to bring the threads of the troubleshooting test lead into contact with the module chassis. This

APPARENT CAUSE OF OCCURRENCE: (cont'd)

created an electrical short between the controller signal common and chassis ground, which produced an erroneous maximum speed demand signal. Flow through the pumps then increased resulting in the low suction pressure and the pump trips.

The test jack (manufactured by Hagan and supplied by Westinghouse) utilizes a conductive threaded sleeve and plastic fastening nut to connect the test jack to the front of the module case with a shouldered fiber washer to electrically insulate the sleeve from the case. A contributing factor to the shorting was lack of insulation between the test jack and the module face plate.

In addition to the loose test jack, investigation revealed approximately 300mV of AC noise on Input 1 of the controller. This was attributed to failure of several electrical capacitors.

ANALYSIS OF OCCURRENCE:

This event did not affect the health and safety of the public. However, it is reportable to the Nuclear Regulatory Commission in accordance with 10CFR 50.73(a)(2)(iv).

In response to the loss of SGFPS, a manual reactor trip was initiated. This was in anticipation to the automatic reactor trip function which would have occurred on Low-Low S/G Level. The Auxiliary Feedwater Pumps (AFPS) {BA} started approximately 13 seconds after the manual reactor trip signal due to the Low-Low S/G Level signal. Engineering is evaluating the present circuit design for AFP automatic start on the sequential loss of the SGFPS.

The plant functioned as designed in response to the loss of both SGFPs and the unit was placed in HOT STANDBY in accordance with procedure. Prior to the SGFPs tripping, feed flow to the steam generators had increased to between 116% and 121% of full flow. This increase in flow resulted in increased condensate flow demand. The increased condensate flow caused the SGFP suction pressure to decrease below the pressure switch setpoints. The SGFPS' low suction trip setpoint is 190 psig (instantaneous) and 215 psig (5 second time delay).

Following the start of the AFPS, the plant experienced an excessive cooldown. This occurs following reactor trips from high power level. In accordance with Emergency Operating Procedure EOP-TRIP-2, a Main Steamline Isolation (an ESF) was initiated stopping the cooldown. The plant was stabilized in Mode 3 utilizing the MS10 atmospheric relief valves to maintain Reactor Coolant System {AB} temperature.

Following the reactor trip, the Pressurizer Heater 22 Backup Group Infeed Circuit Breaker (2EPX) {EC} was placed in service to restore Pressurizer pressure. However, it failed. The 21 Backup Group Heaters were then placed in service and Pressurizer pressure was

ANALYSIS OF OCCURRENCE: (cont'd)

returned to normal. The 2EPX breaker failure did not affect event recovery.

CORRECTIVE ACTION:

The loose SGFP master controller test jack was repaired and other jacks in the SGFP speed control loop circuitry were inspected and repaired as required. The failed capacitors in the master controller module were replaced and the module was returned to service.

The other Hagan protection channel test connections (both Salem Units) are being checked during the protection channel monthly functional surveillance. Several connectors have been found loose and were either repaired or replaced. They would not have resulted in a circuit short. one connector was found in a condition in which a circuit short was possible. It was replaced.

The Hagan process channel test connections, both Salem Units, will be checked for looseness. Repair/replacement will be completed as applicable.

Engineering is reviewing whether changes to the test jacks are needed to prevent event recurrence.

The 2EPX breaker, Asea Brown Boveri Model K-1600, was replaced with a new breaker (same manufacturer and model). Troubleshooting the failed 2EPX breaker revealed the breaker line side fingers and line side bus stabs exhibited heat stress and had partially melted. This is indicative of high resistance between the breaker fingers and bus stabs. Meggering did not reveal shorts or electrical grounds on line and load sides. The Ground Fault Detection System did not detect any grounds. The transient data recorder did not detect any abnormal electrical system current or voltage. Testing on 2XFR2E6DAX, 2EP Pressurizer Heater 4160/480 VAC transformer, did not reveal damage. Failure analysis of the breaker is continuing. Additional corrective action will be based on analysis results.

Engineering is evaluating the present circuit design for AFP automatic start on the sequential loss of the SGFPS.

Plant excessive cooldown, following reactor trips, is being addressed by Engineering.

General Manager -

Salem Operations

MJP: pc
SORC Mtg. 93-019

ATTACHMENT 1 TO 9303040255 PAGE 1 OF 1

PSE&G
Public Service Electric and Gas Company P.O. Box 236
Hancocks Bridge, New Jersey 08038

Salem Generating Station

February 26, 1993

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION
LICENSE NO. DPR-75
DOCKET NO. 50-311
UNIT NO. 2

LICENSEE EVENT REPORT 93-002-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73(a)(2)(iv). This report is required to be issued within thirty (30) days of event discovery.

Sincerely yours,

C. A. Vondra
General Manager -
Salem Operations

MJP: pc

Distribution

The Energy People

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